NON-PUBLIC?: N
ACCESSION #: 9505190131
LICENSEE EVENT REPORT (LER)

FACILITY NAME: SURRY POWER STATION, Unit 1 PAGE: 1 OF 5

DOCKET NUMBER: 05000280

TITLE: Automatic Actuation of Auxiliary Feedwater on Low-Low
Steam Generator Level
EVENT DATE: 04/12/95 LER #: 95-003-00 REPORT DATE: 05/12/95

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 0%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: D. A. Christian, Station Manager TELEPHONE: (804) 357-3184

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: JD COMPONENT: CBL4 MANUFACTURER: C515
REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On April 12, 1995, Unit 1 was operating at 45% reactor power due to a dropped rod and an out of band delta flux. Efforts to correct delta flux caused by the dropped rod were not performed because the same circuit supplying the dropped rod also supplied another control rod in a different control bank. It was believed that movement of this control bank to correct delta flux or as part of an orderly shutdown could have caused a second rod to drop. A decision was made to manually trip the reactor. At 1756 hours, after Operator preparation and briefings, the reactor was manually tripped. Following the trip, the turbine and generator tripped as designed. As expected, all Auxiliary Feedwater pumps automatically started upon receipt of the low-low Steam Generator level signal. At 2120 hours, due to the automatic Engineered Safety Feature (Auxiliary Feedwater) actuation, a four hour non-emergency report was made to the NRC Operations Center in accordance with 10CFR50.72 (b)(2)(ii). The cause of the dropped rod was a cable failure in

containment. The cable supplied power to the stationary coil for Control Rod J-7. This event is being reported as a result of actuation of Engineered Safety Features, pursuant to 10CFR50.73(a)(2)(iv).

END OF ABSTRACT

1.0 DESCRIPTION OF THE EVENT

On April 12, 1995, Unit 1 was operating at 100% reactor power when Control Rod J-7 EIIS-JD! on Control Bank D dropped into the core at 0820 hours. Turbine power was reduced automatically and the control rods inserted automatically as designed. Reactor power stabilized at 73%. The steam dumps opened in response to turbine load reject, and subsequently closed, as designed, based on Tave/Tref mismatch.

In accordance with the abnormal procedure on rod control system malfunction, power was manually reduced to below 70% reactor power. At-power shutdown margin calculations were verified to be within the required Technical Specifications (TS) limit. The High Flux Trip setpoint was reset to 85% and Rod Stop setpoint was reset to 81%, as required by TS.

An investigation of dropped Control Rod J-7 identified blown fuses on the stationary and movable gripper coil. The movable coil circuit for J-7 also supplies the movable coil for Control Rod K-6 which is in Group 2 of Control Bank D. Further movement of the control rods was restricted due to a concern that a second dropped rod could occur if Control Bank D was moved.

At 1158 hours, the delta flux caused by the dropped rod deviated from its target band. Operation of the unit above 50% reactor power with delta flux outside the target band is permitted by TS for a maximum of one hour in any 24-hour period. At 1244 hours, reduction in reactor power to less than 50% was initiated prior to the expiration of the delta flux penalty time. After the unit stabilized, power was subsequently reduced to 45% power to allow for resetting the High Flux Trip setpoints as required by TS. During the ramp down, Main Feedwater (MFW) Pump EIIS-SJ,P! A tripped on low MFW flow due to a malfunction of its recirculation valve which prevented the valve from fully opening.

Operation of the unit continued at less than 50% power as allowed by TS. Efforts to repair the dropped rod or to correct the delta flux by moving control rods were restricted due to the possibility of dropping a second rod. At 1756 hours, delta flux exceeded +16% due to the dropped rod and xenon redistribution. Since the existing problems with the control rod system created the potential for dropping additional rods during an orderly shutdown, a decision was made to trip the reactor. Following Operator preparation and briefings, the reactor was manually tripped.

The turbine and generator tripped as designed. All control rods were verified to be fully inserted into the core. As expected, all Auxiliary Feedwater (AFW) pumps automatically started on low-low Steam Generator (SG) level following the trip. The SGs were initially fed by AFW and later by MFW.

The Reactor Coolant System (RCS) EIIS-AB! initially cooled to 541.5 degrees F following the reactor trip and then experienced a slow cooldown to 538 degrees F (below normal no load temperature). The reactivity shutdown margin was calculated following the RCS cooldown to ensure that TS and administrative shutdown margin limits were satisfied.

At 2120 hours, due to the automatic Engineered Safety Feature (ESF) (Auxiliary Feedwater) actuation, a four hour non-emergency report was made to the NRC Operations Center in accordance with 10CFR50.72 (b)(2)(ii). This event is being reported pursuant to 10 CFR 50.73 (a)(2)(iv) as a result of actuation of Engineered Safety Features.

2.0 SAFETY CONSEQUENCES AND IMPLICATIONS

Following the manual reactor trip, the Reactor Protection System (RPS) EIIS-JC! functioned as designed and all control rods inserted into the core. Appropriate Operator actions were taken in accordance with abnormal and emergency operating procedures to verify the performance of system automatic actions and to respond to abnormal conditions. On the SG low-low level signal, AFW flow initiated, as designed, and provided flow to the SGs. Station Operations personnel placed the plant in a stable, hot shutdown condition. This event resulted in no safety consequences or implications. The health and safety of the public were not affected.

3.0 CAUSE

The cause of the dropped rod was identified as a failed stationary coil conductor for Control Rod J-7, located in the penetration area inside Unit 1 Containment. It was identified that the failed conductor was found in a bundle of Kapton insulated conductors bound together with tape used for mechanical protection. The failed conductor was found to be in contact with a structural member on the floor which abraded the Kapton insulation and exposed bare conductors.

At the direction of station management, the reactor was manually tripped. The decision was based on the inability to correct delta flux after Control Rod J-7 dropped and on the potential of dropping another control rod if the control rod powered from the same movable gripper circuit was moved.

As the Turbine Stop Valves EIIS-SB,ISV! closed following the reactor trip, SG pressure increased and SG levels shrank. On the SG low-low level signal, AFW flow initiated, as designed, and provided flow to the SGs. This ESF actuation was expected and discussed by the Control Room Operators prior to the reactor trip.

4.0 IMMEDIATE CORRECTIVE ACTIONS

Following the trip, Control Room Operators acted promptly to ensure the plant was placed in a safe, hot shutdown condition in accordance with emergency and other operating procedures. The Shift Technical Advisor (STA) calculated the reactivity shutdown margin to ensure that TS and administrative shutdown margin limits were satisfied. The STA also monitored the critical safety function status trees to verify that the unit conditions were acceptable. Plant response was as expected and the unit was stabilized at hot shutdown. A four hour non-emergency report, due to the automatic actuation of ESF (Auxiliary Feedwater), was made in accordance with 10CFR50.72 (b)(2)(ii).

5.0 ADDITIONAL CORRECTIVE ACTIONS

A Containment entry was made to troubleshoot Control Rod J-7. The investigation revealed a failed stationary coil conductor for Control Rod J-7. The stationary conductor along with the stationary and movable coil fuses for Control Rod J-7 were replaced. In addition, four other condu
tors within this bundle were replaced due to degradation of insulation and wires. Inspections of other electrical penetrations in Unit 1 Containment verified no other damage. Control Rod J-7 was returned to service following

satisfactory post maintenance testing.

6.0 ACTIONS TO PREVENT RECURRENCE

A RCE was initiated to determine the cause of the failed conductor. A detailed failure analysis will be completed as part of the Root Cause. Recommendations will be implemented when the RCE is finalized.

Inspections of the containment electrical penetrations in both Unit 1 and Unit 2 will be performed to identify and evaluate similar conditions.

7.0 SIMILAR EVENTS

None

8.0 MANUFACTURER/MODEL NUMBER

None

ATTACHMENT TO 9505190131 PAGE 1 OF 1

10CFR50.73

Virginia Electric and Power Company
Surry Power Station
5570 Hog Island Road
Surry, Virginia 23883

May 12,1995

U. S. Nuclear Regulatory Commission Serial No.: 95-253
Document Control Desk SPS:BAG
Washington, D. C. 20555 Docket No.: 50-280
License No.: DPR-32

Dear Sirs:

Pursuant to Surry Power Station Technical Specifications, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Unit 1.

REPORT NUMBER

50-280/95-003-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Very truly yours,

D. A. Christian
Station Manager

Enclosure

pc: Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

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